NON-PUBLIC?: N
ACCESSION #: 9002090006
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Perry Nuclear Power Plant, Unit 1 PAGE: 1 OF 3

DOCKET NUMBER: 05000440

TITLE: Improper A. C. Bus Transfer Due To Operator Error Results In A Complete Loss Of Feedwater And A Reactor Scram On Low Reactor Water Level
EVENT DATE: 01/07/90 LER #: 90-001-00 REPORT DATE: 02/02/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv) and "OTHER" Technical Specification 3.5.1 Action g.

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On January 7, 1990, at 1132, a reactor scram occurred due to a reactor water level of less than Level 3 (+177.7 inches above the top of active fuel). The low water level occurred after an improper transfer of 480 volt AC power supplies resulted in a momentary power interruption to the feedwater control circuitry and a total loss of feedwater flow.

The cause of this event was Personnel Error, inattention to detail. An operator inadvertently pulled the wrong fuses, which resulted in the 480 volt bus being deenergized when the bus transfer was made.

To prevent recurrence, the operator has been counseled on the importance of paying close attention to detail when operating plant equipment. The system operating instruction was revised and supplemental training on

live-bus transfers was developed. Investigation is being performed into possible design changes to increase Feedwater Control System availability in the event of loss of non-essential busses. As part of the established requalification training program, all plant licensed operators will be instructed on the lessons learned from this event.

Submittal of this report also meets the requirements for Technical Specification 3.5.1 Action g. which requires a Special Report following any Emergency Core Cooling System actuation and injection into the Reactor Coolant System.

END OF ABSTRACT

On January 7, 1990, at 1132, a reactor scram occurred due to reactor pressure vessel RPV! water level less than Level 3 (+177.7 inches above top of active fuel (TAF)). Prior to the event, the plant was in Operational Condition 1 (Power Operation) at approximately 100 percent of rated thermal power with the reactor pressure vessel at saturated conditions at approximately 1025 psig.

On January 7, 1990, at approximately 1100, a non-essential 480 volt bus was being transferred to its alternate power supply, in accordance with System Operating Instruction (SOI-R10) "Plant Electrical System," in preparation for scheduled maintenance. Operators inadvertently removed the wrong control power fuses and upon completion of the transfer, bus F-1-C became completely deenergized. The resulting power interruption to feedwater control circuitry caused a complete loss of feedwater flow as well as a shift of Reactor Recirculation System AD! pumps to slow speed. Both Turbine Driven Feedwater Pumps (TDFP) went to minimum speed. The rapid decrease in reactor water level resulted in a reactor scram at Level 3, at 1132, and Reactor Core Isolation Cooling (RCIC)BN! and High Pressure Core Spray (HPCS) BG! initiation at Level 2 (+129.8 inches above TAF) approximately 5 seconds later. The RCIC initiation signal caused the Main Turbine and both TDFPs to trip which resulted in the Motor Driven Feedwater Pump (MDFP) auto-starting at minimum flow. Minimum RPV level reached during this event was approximately +78 inches above TAF. As a result of the injections from the HPCS and RCIC systems, RPV level was restored to normal operating range at approximately 1136. At 1143, an "Unusual Event" was declared due to HPCS initiation. All off-normal and plant emergency instructions were appropriately implemented. At 1158, HPCS was shutdown to standby readiness and the MDFP was used to control RPV level. The RCIC system isolated, at 1209, due to equipment room high differential temperature with no resultant effect on RPV water level due to MDFP availability (refer to LER 90-002).

At 1216, bus F-1-C was reenergized. The "Unusual Event" was terminated at 1228. This was the fifth HPCS injection cycle to date, including those performed during startup testing activities, and the injection nozzle usage factor is currently less than 0.70. The post scram evaluation was completed, a plan to test the RCIC system differential temperature instrumentation under operating conditions was developed, and the plant entered Operational Condition 2 (Startup) on January 15, 1990, at 2359.

The cause of this event was Personnel Error, inattention to detail. SOI-R10 directs control power fuses to be pulled to disable a tie breaker interlock allowing both the normal and alternate power supply to be tied to bus F-1-C at the same time, permitting bus power to be transferred without interruption. An operator pulled the wrong fuses despite the fuses being clearly labeled and the fuses being clearly identified by the procedure. Because the wrong fuses were pulled, the tie breaker tripped immediately upon closure, removing the alternate power supply from the bus just before the operator disconnected the normal power supply.

A loss of feedwater flow at 100 percent of rated power has been analyzed as discussed in Chapter 15 (Section 15.2.7) of the Perry Updated Safety Analysis Report (USAR). The analyzed transient results in a reactor scram at Level 3, and the initiation of HPCS and RCIC at Level 2. All plant conditions were within the envelope of the USAR analysis, and all plant Technical Specification related systems responded as designed to maintain the plant in a safe condition with the exception of the RCIC isolation.

The RCIC System Isolation is separately addressed in LER 90-002.

Based on the performance of plant equipment in response to the previously analyzed transient, this event is not considered to be safety significant. A similar event has been previously documented by LER 88-012 in which the Reactor scrammed from 100 percent of rated power due to a loss of feedwater caused by personnel error and procedural inadequacy. The corrective actions, which included counseling the operators involved, operator training on the sequence of events, and revision of the applicable system operating instruction, could not have prevented the January 7, 1990 event.

To prevent recurrence, the operator has been counseled on the importance of paying close attention to detail when operating plant equipment. Also, SOI-R10 was revised to have operators ensure that tie breakers stay energized before removing power during live bus transfers. Additionally, a training video tape has been made on live-bus transfers. All on-shift Electrical Operator personnel are being trained to this tape and it was incorporated into the Electrical Operator training program. Also investigation is being performed into possible design changes to increase Feedwater Control system availability in the event of loss of non-essential busses. As part of the established requalification training program, all plant licensed operators will be instructed on the lessons learned from this event.

Submittal of this report also meets the requirements for Technical Specification 3.5.1 Action g, which requires a Special Report following any Emergency Core Cooling System actuation and injection into the Reactor Coolant System.

Energy Industry Identification System Codes are identified in the text as XX!.

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